7590-01-P

NUCLEAR REGULATORY COMMISSION

BIWEEKLY NOTICE

APPLICATIONS AND AMENDMENTS TO FACILITY OPERATING LICENSES

INVOLVING NO SIGNIFICANT HAZARDS CONSIDERATIONS

[NRC-2009-0148]

I. Background

Pursuant to section 189a. (2) of the Atomic Energy Act of 1954, as amended (the Act), the U.S. Nuclear Regulatory Commission (the Commission or NRC staff) is publishing this regular biweekly notice. The Act requires the Commission publish notice of any amendments issued, or proposed to be issued and grants the Commission the authority to issue and make immediately effective any amendment to an operating license upon a determination by the Commission that such amendment involves no significant hazards consideration, notwithstanding the pendency before the Commission of a request for a hearing from any person.

This biweekly notice includes all notices of amendments issued, or proposed to be issued from March 12, 2009 to March 25, 2009. The last biweekly notice was published on March 24, 2009 (74 FR 12390).

NOTICE OF CONSIDERATION OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES, PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION, AND OPPORTUNITY FOR A HEARING

The Commission has made a proposed determination that the following amendment requests involve no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. The basis for this proposed determination for each amendment request is shown below.

The Commission is seeking public comments on this proposed determination. Any comments received within 30 days after the date of publication of this notice will be considered in making any final determination.

Normally, the Commission will not issue the amendment until the expiration of 60 days after the date of publication of this notice. The Commission may issue the license amendment before expiration of the 60-day period provided that its final determination is that the amendment involves no significant hazards consideration. In addition, the Commission may issue the amendment prior to the expiration of the 30-day comment period should circumstances change during the 30-day comment period such that failure to act in a timely way would result, for example in derating or shutdown of the facility. Should the Commission take action prior to the expiration of either the comment period or the notice period, it will publish in the *Federal Register* a notice of issuance. Should the Commission make a final No Significant Hazards Consideration Determination, any hearing will take place after issuance. The Commission expects that the need to take this action will occur very infrequently.

Written comments may be submitted by mail to the Chief, Rulemaking and Directives Branch, TWB-05-B01M, Division of Administrative Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and should cite the publication date and page number of this *Federal Register* notice. Copies of written comments received may be examined at the Commission's Public Document Room (PDR), located at One White Flint North, Public File Area O1F21, 11555 Rockville Pike (first floor), Rockville, Maryland.

Within 60 days after the date of publication of this notice, any person(s) whose interest may be affected by this action may file a request for a hearing and a petition to intervene with respect to issuance of the amendment to the subject facility operating license. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's "Rules of Practice for Domestic Licensing Proceedings" in 10 CFR Part 2. Interested person(s) should consult a current copy of 10 CFR 2.309, which is available at the Commission's PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, http://www.nrc.gov/reading-rm/doc-collections/cfr/. If a request for a hearing or petition for leave to intervene is filed within 60 days, the Commission or a presiding officer designated by the Commission or by the Chief Administrative Judge of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the Chief Administrative Judge of the Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

As required by 10 CFR 2.309, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following general requirements: 1) the name, address, and telephone number of the requestor or petitioner; 2) the nature of the requestor's/petitioner's right under the Act to be made a party to the proceeding; 3) the nature and extent of the requestor's/petitioner's property, financial, or other interest in the proceeding; and 4) the possible effect of any decision or order which may be entered in the proceeding on the requestor's/petitioner's interest. The petition must also set forth the specific contentions which the petitioner/requestor seeks to have litigated at the proceeding.

Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner/requestor shall provide a brief explanation of the bases for the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner/requestor intends to rely in proving the contention at the hearing. The petitioner/requestor must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner/requestor intends to rely to establish those facts or expert opinion. The petition must include sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact. Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the petitioner/requestor to relief. A petitioner/requestor who fails to satisfy these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing.

If a hearing is requested, and the Commission has not made a final determination on the issue of no significant hazards consideration, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to decide when the hearing is held. If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment. If the final determination is that the amendment request involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

All documents filed in NRC adjudicatory proceedings, including a request for hearing, a petition for leave to intervene, any motion or other document filed in the proceeding prior to the submission of a request for hearing or petition to intervene, and documents filed by interested governmental entities participating under 10 CFR 2.315(c), must be filed in accordance with the NRC E-Filing rule, which the NRC promulgated in August 28, 2007 (72 FR 49139). The E-Filing process requires participants to submit and serve all adjudicatory documents over the internet or in some cases to mail copies on electronic storage media. Participants may not submit paper copies of their filings unless they seek a waiver in accordance with the procedures described below.

To comply with the procedural requirements of E-Filing, at least five (5) days prior to the filing deadline, the petitioner/requestor must contact the Office of the Secretary by e-mail at hearingdocket@nrc.gov, or by calling (301) 415-1677, to request (1) a digital ID certificate, which allows the participant (or its counsel or representative) to digitally sign documents and access the E-Submittal server for any proceeding in which it is participating; and/or (2) creation of an electronic docket for the proceeding (even in instances in which the petitioner/requestor (or its counsel or representative) already holds an NRC-issued digital ID certificate). Each petitioner/requestor will need to download the Workplace Forms Viewer™ to access the Electronic Information Exchange (EIE), a component of the E-Filing system. The Workplace Forms Viewer™ is free and is available at http://www.nrc.gov/site-help/e-submittals/install-viewer.html. Information about applying for a digital ID certificate is available on NRC's public website at http://www.nrc.gov/site-help/e-submittals/apply-certificates.html.

Once a petitioner/requestor has obtained a digital ID certificate, had a docket created, and downloaded the EIE viewer, it can then submit a request for hearing or petition for leave to intervene. Submissions should be in Portable Document Format (PDF) in accordance with NRC guidance available on the NRC public website at http://www.nrc.gov/site-help/e-submittals.html. A filing is considered complete at the time the filer submits its documents through EIE. To be timely, an electronic filing must be submitted to the EIE system no later than 11:59 p.m. Eastern Time on the due date. Upon receipt of a transmission, the E-Filing system time-stamps the document and sends the submitter an e-mail notice confirming receipt of the document. The EIE system also distributes an e-mail notice that provides access to the document to the NRC Office of the General Counsel and any others who have advised the Office of the Secretary that they wish to participate in the proceeding, so that the filer need not serve the documents on those participants separately. Therefore, applicants and other participants (or their counsel or representative) must apply for and receive a digital ID certificate before a hearing request/petition to intervene is filed so that they can obtain access to the document via the E-Filing system.

A person filing electronically may seek assistance through the "Contact Us" link located on the NRC website at http://www.nrc.gov/site-help/e-submittals.html or by calling the NRC electronic filing Help Desk, which is available between 8:00 a.m. and 8:00 p.m., Eastern Time, Monday through Friday, excluding government holidays. The help electronic filing Help Desk can be contacted by telephone at 1-866-672-7640 or by e-mail at MSHD.Resource@nrc.gov.

Participants who believe that they have a good cause for not submitting documents electronically must file a motion, in accordance with 10 CFR 2.302(g), with their initial paper filing requesting authorization to continue to submit documents in paper format. Such filings must be submitted by: (1) first class mail addressed to the Office of the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, Attention: Rulemaking and Adjudications Staff; or (2) courier, express mail, or expedited delivery service to the Office of the Secretary, Sixteenth Floor, One White Flint North, 11555 Rockville, Pike, Rockville, Maryland, 20852, Attention: Rulemaking and Adjudications Staff. Participants filing a document in this manner are responsible for serving the document on all other participants. Filing is considered complete by first-class mail as of the time of deposit in the mail, or by courier, express mail, or expedited delivery service upon depositing the document with the provider of the service.

Non-timely requests and/or petitions and contentions will not be entertained absent a determination by the Commission, the presiding officer, or the Atomic Safety and Licensing Board that the petition and/or request should be granted and/or the contentions should be admitted, based on a balancing of the factors specified in 10 CFR 2.309(c)(1)(i)-(viii).

Documents submitted in adjudicatory proceedings will appear in NRC's electronic hearing docket which is available to the public at http://ehd.nrc.gov/EHD_Proceeding/home.asp, unless excluded pursuant to an order of the Commission, an Atomic Safety and Licensing Board, or a Presiding Officer. Participants are requested not to include personal privacy information, such as social security numbers, home addresses, or home phone numbers in their filings, unless an NRC regulation or other law requires submission of such information. With respect to copyrighted works, except for limited excerpts that serve the purpose of the adjudicatory filings and would constitute a Fair Use application, participants are requested not to include copyrighted materials in their submission.

For further details with respect to this amendment action, see the application for amendment which is available for public inspection at the Commission's PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the ADAMS Public Electronic Reading Room on the Internet at the NRC Web site, http://www.nrc.gov/reading-rm/adams.html. If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the PDR Reference staff at 1 (800) 397-4209, (301) 415-4737 or by email to pdr.resource@nrc.gov.

Duke Energy Carolinas, LLC, et al., Docket Nos. 50-413 and 50-414, Catawba Nuclear Station, Units 1 and 2, York County, South Carolina

Date of amendment request: September 2, 2008

Description of amendment request: The amendments would revise the technical specifications to allow manual operation of the containment spray system and to revise the upper and lower limits of the refueling water storage tank.

Basis for proposed no significant hazards consideration determination: As required by Title 10 of the *Code of Federal Regulations* (10 CFR) 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   The Containment Spray System and RWST [refueling water storage tank] are accident mitigation equipment. As such, changes in operation of these systems cannot have an impact on the probability of an accident.

   The RWST will continue to comply with all applicable regulatory requirements and design criteria following approval of the proposed changes (e.g., train separation, redundancy, and single failure). The water level on the containment floor will be higher at the start of transfer to the containment sump but will remain below the maximum design level analyzed for equipment submergence. The change in the sump pH will not result in a significant increase in radiological consequences of a LOCA [loss of coolant accident]. Therefore, the design functions performed by the equipment are not changed.

   The delay in containment spray operation will result in an increase in containment temperature, containment pressure, offsite dose, and control room dose during a LOCA or high energy line break inside containment. Containment analyses have been performed to demonstrate that containment pressure and temperature remain within the design limits and there is no significant impact on the environmental qualification for equipment inside containment. The impact on piping and supports is acceptable without modification. The reduction in fission product removal due to delayed containment spray operation does not result in exceeding the offsite dose and control room dose limits in 10 CFR 50.67 and 10 CFR Part 50, Appendix A, GDC 19. The analysis of the change in containment conditions due to a single failure of an operating spray pump and the suspension of containment spray determined that the pressure remained below the design limits.

Regarding the proposed change to adopt TSTF-493, Rev. 3 on a limited basis, the change clarifies the requirements for instrumentation to ensure the instrumentation will actuate as assumed in the safety analysis. Instruments are not an assumed initiator of any accident previously evaluated. As a result, the proposed change will not increase the probability of an accident previously evaluated. The proposed change will ensure that the instruments actuate as assumed to mitigate the accidents previously evaluated. As a result, the proposed change will not increase the consequences of an accident previously evaluated.

Based on this discussion, the proposed amendment does not significantly increase the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   The modifications to install RWST narrow range level indication will be seismically qualified and isolated from the safety related portion of the RWST level indication system. As such, the new level indication will not create the possibility of a new or different kind of accident.

   The modification to the low level setpoint will not install any new plant equipment. The setpoint will continue to be included within the engineered safeguards features instrumentation and monitored according to the applicable surveillance requirements. The evaluation of the new level setpoint and the change in the swapover sequence concluded that the equipment aligned to the sump will continue to have sufficient suction pressure prior to containment sump suction swapover. The design of the RWST low level instrumentation-complies with all applicable regulatory requirements and design criteria.

   The overall function of the Containment Spray System is not changed by this proposed amendment. The proposed change alters the method of controlling the safety system following a design basis event so that manual actions are substituted for automatic actions. Calculations confirm that these actions will be taken within the appropriate scenario sequence timing to provide containment cooling and source term reduction with no significant increase in radiological consequences and without exceeding containment design limits.

Regarding the proposed change to adopt TSTF-493, Rev. 3 on a limited basis, the change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The change does not alter assumptions made in the safety analysis but ensures that the instruments behave as assumed in the accident analysis. The proposed change is consistent with the safety analysis assumptions.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed amendment involve a significant reduction in the margin of safety?

Response: No

The proposed change has the potential to increase the radiological dose at the site boundary and in the control room. However, the calculations demonstrate that the dose consequences at the site boundary, low population zone, and control room remain within regulatory acceptance limits. Additional analysis concluded:

- Peak containment pressure for analyzed design basis accidents will not be significantly increased and containment design limits will not be exceeded.
- Assumptions used in the environmental qualification of equipment exposed to the containment atmosphere remain bounding.
- Pumps aligned to the RWST and to the containment sump will have adequate suction pressure.

Regarding the proposed change to adopt TSTF-493, Rev. 3 on a limited basis, the change clarifies the requirements for instrumentation to ensure the instrumentation will actuate as assumed in the accident analysis. No change is made to the accident analysis assumptions and no margin of safety is reduced as part of this change.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Ms. Lisa F. Vaughn, Associate General Counsel and Managing Attorney, Duke Energy Carolinas, LLC, 526 South Church Street, EC07H, Charlotte, NC 28202

NRC Branch Chief: Melanie C. Wong

Duke Energy Carolinas, LLC, et al., Docket Nos. 50-413 and 50-414, Catawba Nuclear Station, Units 1 and 2, York County, South Carolina

Date of amendment request: October 2, 2008

Description of amendment request: The amendments would revise Technical Specifications (TS) associated with the verification of ice condenser door operability. The proposed amendment affects the current TS surveillance requirements 3.6.13.5 and 3.6.13.6.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The only analyzed accidents of possible consideration in regards to changes potentially affecting the ice condenser are a loss of coolant accident (LOCA) and a high energy line break (HELB) inside Containment. However, the ice condenser is not postulated as being the initiator of any LOCA or HELB. This is because it is designed to remain functional following a design basis earthquake, and the ice condenser does not interconnect or interact with any systems that interconnect or interact with the Reactor Coolant or Main Steam Systems. Since these proposed changes do not result in, or require, any physical change to the ice condenser that could introduce an interaction with the Reactor Coolant or Main Steam Systems, then there can be no change in the probability of an accident previously evaluated. Regarding consequences of analyzed accidents, the ice condenser is an engineered safety feature designed, in part, to limit the Containment sub-compartment and Containment vessel pressure immediately following the initiation of a LOCA or HELB. Conservative sub-compartment and Containment pressure analysis shows these criteria will be met if the total ice mass within the ice bed is maintained in accordance with the DBA analysis; therefore, the proposed TS [Technical Specification] SR [surveillance requirement] changes of these requirements will not increase the consequences of any accident previously evaluated.

Thus, based on the above, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   As previously described, the ice condenser is not postulated as being the initiator of any design basis accident. The proposed changes do not impact any plant system, structure or component that is an accident initiator. The proposed TSs and TS Bases changes do not involve any hardware changes to the ice condenser or other change that could create any new accident mechanisms. Therefore, there can be no new or different accidents created from those already identified and evaluated

3. Does the proposed amendment involve a significant reduction in the margin of safety?

   Response: No.

   Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following an accident situation. These barriers include the fuel cladding, the reactor coolant system, and the Containment system. The performance of the fuel cladding and the reactor coolant system will not be impacted by the proposed changes. The Application provides a description of additional sub-compartment and Containment pressure response analysis that has been performed. This analysis demonstrates that Containment will remain fully capable of performing its design function with implementation of the proposed changes. Therefore, no safety margin will be significantly impacted.

> The changes proposed in this LAR [license amendment request] do not make any physical alteration to the ice condenser doors, nor does it affect the required functional capability of the doors in any way. The intent of the proposed changes to the ice condenser door surveillance requirements is to eliminate an unnecessary and overly restrictive Lower Inlet Door torque surveillance test. There will be no degradation in the operable status of the ice condenser doors and the ability to confirm operability for the ice condenser doors will be maintained, such that the doors will continue to fully perform their safety function as assumed in the plant's safety analyses.
>
> Thus, it can be concluded that the proposed TS and TS Bases changes do not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Ms. Lisa F. Vaughn, Associate General Counsel and Managing Attorney, Duke Energy Carolinas, LLC, 526 South Church Street, EC07H, Charlotte, NC 28202

NRC Branch Chief: Melanie C. Wong

Duke Energy Carolinas, LLC, et al., Docket Nos. 50-413 and 50-414, Catawba Nuclear Station, Units 1 and 2, York County, South Carolina

Date of amendment request: October 8, 2008

Description of amendment request: The amendments would revise the Technical Specifications (TSs) by removing and updating portions of the TSs which are out of date or are obsolete including footnotes and references.

<u>Basis for proposed no significant hazards consideration determination</u>: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed changes are administrative in nature and therefore they do not involve any change in the design, configuration, or operation of the nuclear units. All Limiting Conditions for Operation, Limiting Safety System Settings and Safety Limits specified in the Technical Specifications remain unchanged. The Physical Security and related plans, Operator Training and Requalification Programs, Quality Assurance Programs, and the Emergency Plans will not be materially changed by the proposed license amendment due to its administrative nature.

   The technical qualifications of the operating licensee will not be reduced. Personnel engaged in operation, maintenance, engineering, assessment, training, and other related services will not be changed. Duke officers and executives currently responsible for the overall safe operation of the nuclear plants are expected to continue in the same capacity.

   Therefore, the proposed amendment does not involve an increase in the probability or consequences of an accident previously analyzed.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed changes are administrative in nature and therefore they do not involve any change in the design, configuration, or operation of the nuclear plant. The current plant safety analyses, therefore, remain complete and accurate in addressing the design basis events and in analyzing plant response and consequences.

   The Limiting Conditions for Operations, Limiting Safety System Settings and Safety Limits specified in the Technical Specifications are not affected by the proposed changes. As such, the plant conditions for which the design basis accident analyses were performed remain valid.

The amendment does not introduce a new mode of plant operation or new accident precursors, does not involve any physical alterations to plant configurations or make changes to system set points that could initiate a new or different kind of accident.

Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No.

The proposed changes are administrative in nature and therefore they do not involve a change in the design, configuration, or operation of the nuclear plants. The change does not affect either the way in which the plant, structures, systems, and components perform their safety function or their design and licensing bases.

Plant safety margins are established through Limiting Conditions for Operation, Limiting Safety System Settings and Safety Limits specified in the Technical Specifications. Because there is no change to the physical design of the plant, there is no change to any of these margins.

Therefore, the proposed amendment does not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Ms. Lisa F. Vaughn, Associate General Counsel and Managing Attorney, Duke Energy Carolinas, LLC, 526 South Church Street, EC07H, Charlotte, NC 28202

NRC Branch Chief: Melanie C. Wong

Duke Energy Carolinas, LLC, et al., Docket Nos. 50-413 and 50-414, Catawba Nuclear Station, Units 1 and 2, York County, South Carolina

Date of amendment request: October 14, 2008

Description of amendment request: The amendments would revise the Technical Specification [TS] Administrative Controls, "Inservice Testing Program," for consistency with the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(f)(4) for pumps and valves which are classified as American Society of Mechanical Engineers [ASME] Code Class 1, Class 2, and Class 3.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   The proposed changes revise TS 5.5.8, "Inservice Testing Program," for consistency with the requirements of 10 CFR 50.55a(f)(4) regarding the inservice testing of pumps and valves which are classified as ASME Code Class 1, Class 2, and Class 3. The proposed changes incorporate revisions to the ASME Code that result in a net improvement in the measures for testing pumps and valves.

   The proposed changes do not impact any accident initiators or analyzed events or assumed mitigation of accident or transient events. The proposed change does not involve the addition or removal of any equipment, or any design changes to the facility. Therefore, these proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

The proposed changes revise TS 5.5.8, "Inservice Testing Program," for consistency with the requirements of 10 CFR 50.55a(f)(4) regarding the inservice testing of pumps and valves which are classified as ASME Code Class 1, Class 2, and Class 3. The proposed changes incorporate revisions to the ASME Code that result in a net improvement in the measures for testing pumps and valves.

The proposed changes do not involve a modification to the physical configuration of the plant nor does it involve a change in the methods governing normal plant operation. The proposed changes will not impose any new or different requirements or introduce a new accident initiator, accident precursor, or malfunction mechanism. Additionally, there is no change in the types or increases in the amounts of any effluent that may be released offsite and there is no increase in individual or cumulative occupational exposure. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No

The proposed changes revise TS 5.5.8, "Inservice Testing Program," for consistency with the requirements of 10 CFR 50.55a(f)(4) regarding the inservice testing of pumps and valves which are classified as ASME Code Class 1, Class 2, and Class 3. The proposed changes do not involve a modification to the physical configuration of the plant nor does it change the methods governing normal plant operation. The proposed changes incorporate revisions to the ASME Code that result in a net improvement in the measures for testing pumps and valves. The safety function of the affected pumps and valves will be maintained. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

<u>Attorney for licensee</u>: Ms. Lisa F. Vaughn, Associate General Counsel and Managing Attorney, Duke Energy Carolinas, LLC, 526 South Church Street, EC07H, Charlotte, NC 28202

NRC Branch Chief: Melanie C. Wong

Duke Energy Carolinas, LLC, Docket Nos. 50-369 and 50-370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina

Date of amendment request: October 2, 2008

Description of amendment request: The proposed amendments would revise technical specifications (TS) associated with the verification of ice condenser door operability. The proposed amendment affects the current TS surveillance requirements 3.6.13.5 and 3.6.13.6.

Basis for proposed no significant hazards consideration determination: As required by Title 10 of the *Code of Federal Regulations* (10 CFR) 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The only analyzed accidents of possible consideration in regards to changes potentially affecting the ice condenser are a loss of coolant accident (LOCA) and a high energy line break (HELB) inside Containment. However, the ice condenser is not postulated as being the initiator of any LOCA or HELB. This is because it is designed to remain functional following a design basis earthquake, and the ice condenser does not interconnect or interact with any systems that interconnect or interact with the Reactor Coolant or Main Steam Systems. Since these proposed changes do not result in, or require, any physical change to the ice condenser that could introduce an interaction with the Reactor Coolant or Main Steam Systems, then there can be no change in the probability of an accident previously evaluated. Regarding consequences of analyzed accidents, the ice condenser is an engineered safety feature designed, in part, to limit the Containment sub-compartment and Containment vessel pressure immediately following the initiation of a LOCA or HELB. Conservative sub-compartment and Containment pressure analysis shows these criteria will be met if the total ice mass within the ice bed is maintained in accordance with the DBA analysis; therefore, the proposed TS [technical specification] SR [surveillance requirement] changes of these requirements will not increase the consequences of any accident previously evaluated.

Thus, based on the above, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   As previously described, the ice condenser is not postulated as being the initiator of any design basis accident. The proposed changes do not impact any plant system, structure or component that is an accident initiator. The proposed TSs and TS Bases changes do not involve any hardware changes to the ice condenser or other change that could create any new accident mechanisms. Therefore, there can be no new or different accidents created from those already identified and evaluated.

3. Does the proposed amendment involve a significant reduction in the margin of safety?

   Response: No.

   Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following an accident situation. These barriers include the fuel cladding, the reactor coolant system, and the Containment system. The performance of the fuel cladding and the reactor coolant system will not be impacted by the proposed changes. The Application provides a description of additional sub-compartment and Containment pressure response analysis that has been performed. This analysis demonstrates that Containment will remain fully capable of performing its design function with implementation of the proposed changes. Therefore, no safety margin will be significantly impacted.

   The changes proposed in this LAR [license amendment request] do not make any physical alteration to the ice condenser doors, nor does it affect the required functional capability of the doors in any way. The intent of the proposed changes to the ice condenser door surveillance requirements is to eliminate an unnecessary and overly restrictive Lower Inlet Door torque surveillance test. There will be no degradation in the operable status of the ice condenser doors and the ability to confirm operability for the ice condenser doors will be maintained, such that the doors will continue to fully perform their safety function as assumed in the plant's safety analyses.

> Thus, it can be concluded that the proposed TS and TS Bases changes do not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Ms. Lisa F. Vaughn, Associate General Counsel and Managing Attorney, Duke Energy Carolinas, LLC, 526 South Church Street, EC07H, Charlotte, NC 28202

NRC Branch Chief: Melanie Wong

Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc., Docket No. 50-271, Vermont Yankee Nuclear Power Station, Vernon, Vermont

Date of amendment request: February 24, 2009

Description of amendment request: The proposed amendment would revise the Technical Specification (TS) Surveillance Requirement (SR) that governs operability testing of the pressure suppression chamber-drywell vacuum breakers to incorporate the SR contained within the Standard Technical Specifications (STS), NUREG-1433 and delete the SR that requires inspection of the pressure suppression chamber-drywell vacuum breakers. Periodic inspections of the pressure suppression chamber-drywell vacuum breakers are not required by the STS.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration which is presented below:

1. The operation of Vermont Yankee Nuclear Power Station (VY) in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed amendment does not impact the operability of any structure, system or component that affects the probability of an accident or that supports mitigation of an accident previously evaluated. The proposed amendment does not affect reactor operations or accident analysis and has no radiological consequences. The operability requirements for accident mitigation systems remain consistent with the licensing and design basis. Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The operation of VY in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed amendment does not change the design or function of any component or system. No new modes of failure or initiating events are being introduced. Therefore, operation of VY in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The operation of VY in accordance with the proposed amendment will not involve a significant reduction in a margin of safety.

   The proposed amendment does not change the design or function of any component or system. The proposed amendment does not involve any safety limits, safety settings or safety margins. The ability of the pressure suppression chamber-drywell vacuum breakers to perform its intended function will continue to be required in accordance with the VY Technical Specifications.

   Since the proposed controls are adequate to ensure the operability of the pressure suppression chamber-drywell vacuum breakers, there will still be high assurance that the components are operable and capable of performing their respective functions. Therefore, operation of VY in accordance with the proposed amendment will not involve a significant reduction in the margin to safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Mr. William C. Dennis, Assistant General Counsel, Entergy Nuclear Operations, Inc., 400 Hamilton Avenue, White Plains, NY 10601

NRC Branch Chief: Mark G. Kowal

Exelon Generation Company, LLC, Docket Nos. 50-373 and 50-374, LaSalle County Station, Units 1 and 2, LaSalle County, Illinois

Date of amendment request: October 23, 2008

Description of amendment request: The proposed amendments would revise the Technical Specifications (TS) to support the application of alternative source term (AST) methodology with respect to the loss-of-coolant accident and the fuel handling accident. The proposed request is to support a full-scope application of an AST methodology, with the exception that Technical Information Document (TID)-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," will continue to be used as the radiation dose basis for equipment qualification.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   The implementation of AST assumptions has been evaluated in revisions to the analyses of the following limiting design basis accidents at LSCS [LaSalle County Station]:

   - Loss-of-Coolant Accident, and
   - Fuel Handling Accident

Based upon the results of these analyses, it has been demonstrated that, with the requested changes, the dose consequences of these limiting events are within the regulatory requirements and guidance provided by the NRC for use with AST. The regulatory requirements and guidance is presented in 10 CFR 50.67, “Accident source term,” and associated NRC Regulatory Guide 1.183 and Standard Review Plan section 15.0.1. The AST is an input to calculations used to evaluate the consequences of an accident, and does not by itself affect the plant response, or the actual pathway of the radiation released from the fuel. It does, however, better represent the physical characteristics of the release, so that appropriate mitigation techniques may be applied. Therefore, the consequences of an accident previously evaluated are not significantly increased.

The equipment affected by the proposed change is mitigative in nature, and relied upon after an accident has been initiated. Application of the AST does not involve any physical changes to the TS, while they revise certain performance requirements, do not involve any physical modifications to the plant. As a result, the proposed changes do not affect any of the parameters or conditions that could contribute to the initiation of any accidents. As such, removal of operability requirements during the specified conditions will not significantly increase the probability of occurrence for an accident previously analyzed. Since plant design basis accidents initiators are not being altered by adoption of the AST analyses, the probability of an accident previously evaluated is not affected.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed and there are no physical modifications to existing equipment associated with the proposed change). Similarly, it does not physically change any structures, systems, or components involved in the mitigation of any accidents. Thus, no new initiators or precursors of a new or different kind of accident are created.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No

   Safety margins and analytical conservatisms have been evaluated and have been found to be acceptable. The analyzed events have been carefully selected and margin has been retained to ensure that the analyses adequately bound postulated event scenarios. The dose consequences due to design basis accidents comply with the requirements of 10 CFR 50.67 and guidance of Regulatory Guide 1.183.

The proposed change is associated with the implementation of a new licensing basis for LSCS design basis accidents. Approval of the change from the original source term to a new source term taken from Regulatory Guide 1.183 is being requested. The results of the accident analyses, revised in support of the proposed license amendment, are subject to revised acceptance criteria. The analyses have been performed using conservative methodologies, as specified in Regulatory Guide 1.183. Safety margins have been evaluated and analytical conservatism has been utilized to ensure that the analyses adequately bound the postulated limiting event scenario. The dose consequences of these design basis accidents remain within the acceptance criteria presented in 10 CFR 50.67 and Regulatory Guide 1.183.

The proposed change continues to ensure that the doses at the exclusion area boundary and low population zone boundary, as well as the control room, are within corresponding regulatory limits.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the requested amendments involve no significant hazards consideration.

Attorney for licensee: Mr. Bradley J. Fewell, Associate General Counsel, Exelon Nuclear, 4300 Winfield Road, Warrenville, IL 60555.

NRC Branch Chief: Russell Gibbs

Luminant Generation Company LLC, Docket Nos. 50-445 and 50-446, Comanche Peak Steam Electric Station, Units 1 and 2, Somervell County, Texas

Date of amendment request: February 11, 2009

Brief description of amendment: The proposed amendment consists of administrative revision to the operating licenses and Technical Specifications (TSs) to revise the station name from Comanche Peak Steam Electric Station (CPSES) to Comanche Peak Nuclear Power Plant (CPNPP); remove the Table of Contents from TSs and maintain and revise it in accordance with plant administrative procedures; delete TSs 3.2.1.1, 3.2.3.1, 5.5.9.1, 5.6.10 and several footnotes from Tables 3.3.1-1, 3.3.2-1, and TS 3.4.10 since these TSs and footnotes are no longer applicable to CPSES, Units 1 and 2 operation; delete several topical reports from the list of approved analytical methods used to determine core operating limits in TS 5.6.5, no longer in use, since these topical reports have been replaced by standard Westinghouse methods and Westinghouse methods have been approved for use at CPSES, Units 1 and 2, under a separate amendment request; make editorial corrections; and reprint and reissue the entire TS due to adoption of ‘FrameMaker’ software in place of ‘Microsoft Word’ software.

<u>Basis for proposed no significant hazards consideration determination</u>: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   The proposed change revises the station name, removes the Table of Contents from the Technical Specifications, deletes several Technical Specifications and footnotes which are no longer applicable to [CPSES] Unit 1 or Unit 2 operation, renumbers subsequent Technical Specifications, deletes several topical reports from the list of approved analytical methods used to determine core operating limits, and corrects various editorial and formatting errors. The Table of Contents does not include information required by 10 CFR 50.36 [Title 10 of the *Code of Federal Regulations*, Section 50.36] to be reviewed by the NRC [U.S. Nuclear Regulatory Commission] staff and is not required by the regulation. The Technical Specifications and footnotes which are being deleted were only applicable during previous operational cycles and are now defunct requirements since both Units have completed the applicable operational cycles. The topical reports deleted from Technical Specification 5.6.5b are no longer used to determine the core operating limits for Comanche Peak Nuclear Power Plant. The remaining topical reports listed in Technical Specification 5.6.5b will be used to determine the core operating limits for both Comanche Peak Nuclear Power Plant units. All other changes proposed are corrections of previous inadvertent editorial errors or changes in format to increase conformity with the guidelines described in TSTF-RPT-01, "Writer's Guide for Plant-Specific Improved Technical Specifications", published in June, 2005. All of the proposed changes are administrative changes which do not change the meaning, intent, interpretation, or application of the Technical Specifications. None of the proposed changes affect the operation, physical configuration, or function of plant equipment or systems. The changes do not affect the initiators or assumptions of analyzed events; nor do they impact the mitigation of accidents or transient events. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Do the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No

The proposed change revises the station name, removes the Table of Contents from the Technical Specifications, deletes several Technical Specifications and footnotes which are no longer applicable to [CPSES,] Unit 1 or Unit 2 operation, renumbers subsequent Technical Specifications, deletes several topical reports from the list of approved analytical methods used to determine core operating limits, and corrects various editorial and formatting errors. The Table of Contents does not include information required by 10 CFR 50.36 to be reviewed by the Nuclear Regulatory Commission staff and is not required by the regulation. The Technical Specifications and footnotes which are being deleted were only applicable during previous operational cycles and are now defunct requirements since both Units have completed the applicable operational cycles. The topical reports deleted from Technical Specification 5.6.5b are no longer used to determine the core operating limits for Comanche Peak Nuclear Power Plant. The remaining topical reports listed in Technical Specification 5.6.5b will be used to determine the core operating limits for both Comanche Peak Nuclear Power Plant units. All other changes proposed are corrections of previous inadvertent editorial errors or changes in format to increase conformity with the guidelines described in TSTF-RPT-01, "Writer's Guide for Plant-Specific Improved Technical Specifications", published in June, 2005. All of the proposed changes are administrative changes which do not change the meaning, intent, interpretation, or application of the

Technical Specifications. None of the changes alter the plant configuration, require installation of new equipment, alter assumptions about previously analyzed accidents, or impact the operation or function of any plant equipment or systems. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

3. Do the proposed changes involve a significant reduction in a margin of safety?

Response: No

The proposed change revises the station name, removes the Table of Contents from the Technical Specifications, deletes several Technical Specifications and footnotes which are no longer applicable to [CPSES,] Unit 1 or Unit 2 operation, renumbers subsequent Technical Specifications, deletes several topical reports from the list of approved analytical methods used to determine core operating limits, and corrects various editorial and formatting errors. The Table of Contents does not include information required by 10 CFR 50.36 to be reviewed by the Nuclear Regulatory Commission staff and is not required by the regulation. The Technical Specifications and footnotes which are being deleted were only applicable during previous operational cycles and are now defunct requirements since both Units have completed the applicable operational cycles. The topical reports deleted from Technical Specification 5.6.5b are no longer used to determine the core operating limits for Comanche Peak Nuclear Power Plant. The remaining topical reports listed in Technical Specification 5.6.5b will be used to determine the core operating limits for both Comanche Peak Nuclear Power Plant units. All other changes proposed are corrections of previous inadvertent editorial errors or changes in format to increase conformity with the guidelines described in TSTF-RPT-01, “Writer’s Guide for Plant-Specific Improved Technical Specifications”, published in June, 2005. All of the proposed changes are administrative changes which do not change the meaning, intent, interpretation, or application of the Technical Specifications. None of the proposed changes alter the effective technical content of the Technical Specifications. Therefore the proposed changes do not involve a reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Timothy P. Matthews, Esq., Morgan, Lewis and Bockius, 1800 M Street, NW., Washington, DC 20036

NRC Branch Chief: Michael T. Markley

Omaha Public Power District, Docket No. 50-285, Fort Calhoun Station, Unit No. 1, Washington County, Nebraska

Date of amendment request: October 31, 2008

Description of amendment request: The proposed amendment modifies the surveillance requirements in Technical Specification (TS) 3.6(3), "Containment Recirculating Air Cooling and Filtering System," and removes the license conditions related to the replacement and testing of containment air cooling and filtering (CACF) unit high-efficiency particulate air (HEPA) filters and surveillance testing of the CACF unit relief ports. These license conditions were committed to by the licensee in its letter dated April 10, 2008 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML081010122), and implemented via TS Amendment No. 255 (ADAMS Accession No. ML081140390), dated May 2, 2008.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   The containment air cooling and filtering system (CACFS) is not an initiator of any accident previously evaluated at the Fort Calhoun Station (FCS). The CACFS is an accident mitigation system. The design basis function of the CACFS is to limit the containment pressure rise by providing a means for cooling the containment following a loss-of-coolant accident (LOCA) or main steam line break (MSLB). In accordance with TS Amendment No. 255, the CACFS high efficiency particulate air (HEPA) filters are also credited to reduce post-LOCA radioactive leakage from containment.

The proposed changes provide additional assurance that the CACFS is capable of performing its design and licensing basis functions to mitigate these design basis accidents (DBAs). The CACFS face and bypass dampers are aligned to their accident positions permanently causing the CACFS to operate in filtered air mode. Surveillance testing has shown that operating the system in this alignment over long periods does not jeopardize filter performance. Over the lifetime of the plant, the differential pressures measured across the combined HEPA and charcoal filter banks have met test acceptance criteria.

Increasing the number of surveillance requirements will not adversely affect the function of the CACFS but rather provides additional assurance that the CACFS is capable of responding to a DBA.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   The CACFS was designed to remove heat released to the containment atmosphere during a DBA to the extent necessary to maintain the containment structure below its design pressure. The containment airflow continually passes through the cooling coils. The proposed changes to the surveillance requirements do not affect the active function of the CACFS.

   The CACFS will continue to operate in normal and accident conditions to remove heat and radioactive particulates and aerosols. The proposed changes enhance surveillance testing of the CACFS by requiring more frequent exercising of the fans, imposing a more stringent pressure drop limit, specifying a HEPA filter replacement interval, and instituting a requirement to exercise the relief ports. These changes ensure that the CACFS is capable of long-term operation in filtered air mode while remaining capable of providing cooling and filtering sufficient to mitigate design basis accidents.

No credible new failure mechanisms, malfunctions, or accident initiators not previously considered in the design and licensing basis are created and none of the initial condition assumptions of any accident evaluated in the safety analysis are impacted.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No

The containment building and associated penetrations are designed to withstand an internal pressure of 60 psig [pounds per square inch gauge] at 305°F, including all thermal loads resulting from the temperature associated with this pressure, with a leakage rate of 0.1 percent by weight or less of the contained volume per 24 hours. [Omaha Public Power District] credits the CACFS in the containment pressure analysis for a LOCA, and for the containment pressure response to a main steam line break (MSLB).

The proposed changes impose more stringent surveillance test requirements. This provides additional assurance that the CACFS will perform its design basis and licensing basis functions to be capable of long-term post-DBA operation in filtered air mode to limit the containment temperature and pressure increase to within design limits and to reduce post-LOCA radioactive leakage from containment.

Neither the design basis nor the licensing basis for post-DBA containment heat removal is adversely affected by the proposed changes. The ability to maintain design limits for containment peak pressure and temperature, as well as long-term containment pressure and temperature, are preserved.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: David A. Repka, Esq., Winston & Strawn, 1700 K Street, N.W., Washington, DC 20006-3817

NRC Branch Chief: Michael T. Markley

Omaha Public Power District, Docket No. 50-285, Fort Calhoun Station, Unit No. 1, Washington County, Nebraska

Date of amendment request: January 30, 2009

Description of amendment request: The proposed amendment would modify the Fort Calhoun Station (FCS), Unit No. 1, Renewed Operating License No. DPR-40, by adding operability and surveillance testing requirements to the FCS Technical Specifications (TS) for the steam generator (SG) blowdown isolation on a reactor trip. Specifically, The proposed changes will revise TS Limiting Conditions for Operation (LCO) 2.15, *Instrumentation and Control Systems,* Table 2-4, *Instrument Operating Conditions for Isolation Functions,* to include operability requirements for SG blowdown isolation on a reactor trip and to add applicable footnotes. In addition, TS 3.1, *Instrumentation and Control,* Table 3-2, *Minimum Frequencies for Checks, Calibrations and Testing of Engineered Safety Features, Instrumentation and Controls,* is being revised to include the surveillance test requirements for SG blowdown isolation on a reactor trip. An administrative change is also being made to TS LCO 2.15(1), to delete the words "key operated" as the "key" associated with the bypass switches is not a critical element in controlling the use of bypass switches. This amendment will allow FCS to credit an automatic SG blowdown isolation interlock being installed during the 2009 Refueling Outage (RFO).

<u>Basis for proposed no significant hazards consideration determination</u>: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed change provides Technical Specification (TS) operability and surveillance testing requirements for automatic steam generator (SG) blowdown isolation on a reactor trip in the event of a loss of main feedwater (LMFW). Automatic isolation will ensure that the existing 15-minute requirement in the Updated Safety Analysis Report (USAR) Chapter 14.10 safety analysis is met without the risk that an unanticipated distraction could prevent manual action from occurring at the proper time. The installation of this feature will eliminate the need for manual isolation of blowdown and thus will eliminate the associated operator challenge.

   Automatic isolation of blowdown will reduce the consequences of the LMFW event by providing automatic isolation prior to manual isolation being initiated by the operators. Automatic isolation at the time of reactor trip will reduce the severity of the LMFW event by isolating the SGs earlier in the event, thereby conserving SG inventory.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   No new malfunctions are being introduced by this activity, and based on the current redundancy in the design, there are no malfunctions of the SG blowdown isolation valves that challenge nuclear safety.

   The SG blowdown isolation valves will continue to function as currently credited for the LMFW event; thus, this proposed change does not alter their ability to function as containment isolation valves to maintain containment integrity. The manual isolation capability remains unchanged.

A failure analysis has been prepared which shows that the addition of the automatic isolation feature does not introduce a new failure mode or malfunction to the valve circuits. An isolation of SG blowdown, either through the designed circuit following a reactor trip, or during normal operations, does not present a nuclear safety challenge. The capability exists for operators to bypass the isolation signal and restore blowdown as plant conditions warrant.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No.

The addition of an automatic isolation interlock to the SG blowdown isolation valve circuits that close the valves on a reactor trip actually increases the margin of safety by isolating the SG early in the event to maintain SG inventories.

A reactor trip signal is generated in the first seconds of an LMFW due to reduced SG inventories. Because it is desirable to isolate blowdown as soon as possible following the LMFW event, for maximum margin, a reactor trip signal will be used for the SG blowdown isolation interlock. Isolating blowdown earlier in an event provides greater operating margin in terms of maximizing SG inventories. More margin allows operators more time to address operator demands that occur during transient events.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: David A. Repka, Esq., Winston & Strawn, 1700 K Street, N.W., Washington, DC 20006-3817

NRC Branch Chief: Michael T. Markley

Omaha Public Power District, Docket No. 50-285, Fort Calhoun Station, Unit No. 1, Washington County, Nebraska

Date of amendment request: January 30, 2009

Description of amendment request: The proposed amendment would delete those portions of the Technical Specifications (TS) superseded by Title 10 of the *Code of Federal Regulations* (10 CFR) Part 26, Subpart I. The licensee is proposing to adopt the approved Technical Specification Task Force (TSTF) change traveler TSTF-511, Revision 0, "Eliminate Working Hour Restrictions from TS 5.2.2 to Support Compliance with 10 CFR Part 26."

The NRC staff issued a “Notice of Availability of Model Safety Evaluation, Model No Significant Hazards Determination, and Model Application for Licensees That Wish To Adopt TSTF–511, Revision 0, “Eliminate Working Hour Restrictions From TS 5.2.2 To Support Compliance With 10 CFR Part 26,” in the *Federal Register* on December 30, 2008 (73 FR 79923). The notice included a model safety evaluation, a model no significant hazards consideration (NSHC) determination, and a model license amendment request, using the consolidated line item improvement process. In its application dated January 30, 2009, the licensee affirmed the applicability of the model NSHC determination, which is presented below.

Basis for proposed (NSHC) determination: As required by 10 CFR 50.91(a), an analysis of the issue of NSHC determination is presented below:

> Criterion 1--The Proposed Change Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated
>
> The proposed change removes Technical Specification restrictions on working hours for personnel who perform safety related functions. The Technical Specification restrictions are superseded by the worker fatigue

requirements in 10 CFR Part 26. Removal of the Technical Specification requirements will be performed concurrently with the implementation of the 10 CFR Part 26, Subpart I, requirements. The proposed change does not impact the physical configuration or function of plant structures, systems, or components (SSCs) or the manner in which SSCs are operated, maintained, modified, tested, or inspected. Worker fatigue is not an initiator of any accident previously evaluated. Worker fatigue is not an assumption in the consequence mitigation of any accident previously evaluated. Therefore, it is concluded that this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Criterion 2--The Proposed Change Does Not Create the Possibility of a New or Different Kind of Accident from Any Accident Previously Evaluated

The proposed change removes Technical Specification restrictions on working hours for personnel who perform safety related functions. The Technical Specification restrictions are superseded by the worker fatigue requirements in 10 CFR Part 26. Working hours will continue to be controlled in accordance with NRC requirements. The new rule allows for deviations from controls to mitigate or prevent a condition adverse to safety or as necessary to maintain the security of the facility. This ensures that the new rule will not unnecessarily restrict working hours and thereby create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not alter the plant configuration, require new plant equipment to be installed, alter accident analysis assumptions, add any initiators, or effect the function of plant systems or the manner in which systems are operated, maintained, modified, tested, or inspected. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

Criterion 3--The Proposed Change Does Not Involve a Significant Reduction in a Margin of Safety

The proposed change removes Technical Specification restrictions on working hours for personnel who perform safety related functions. The Technical Specification restrictions are superseded by the worker fatigue requirements in 10 CFR Part 26. The proposed change does not involve any physical changes to plant or alter the manner in which plant systems are operated, maintained, modified, tested, or inspected. The proposed change does not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined. The safety analysis acceptance criteria are not affected by this change. The proposed change will not result in plant operation in a configuration outside the design basis. The proposed change does not adversely affect systems that respond to safely shutdown the plant and to maintain the plant in a safe shutdown condition. Removal of plant-specific Technical Specification administrative requirements will not reduce a margin of safety because the requirements in 10 CFR Part 26 are adequate to ensure that worker fatigue is managed.

> Therefore, the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the analysis adopted by the licensee and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves NSHC.

Attorney for licensee: David A. Repka, Esq., Winston & Strawn, 1700 K Street, N.W., Washington, DC 20006-3817

NRC Branch Chief: Michael T. Markley

R.E. Ginna Nuclear Power Plant, LLC, Docket No. 50-244, R.E. Ginna Nuclear Power Plant, Wayne County, New York

Date of amendment request: December 19, 2008

Description of amendment request: The proposed amendment would modify the Technical Specifications (TSs) to (1) correct an error in TS Table 3.3.2-1, "Engineered Safety Feature Actuation System Instrumentation," Function 1.a, to reflect the correct CONDITIONS for applicable Modes 1, 2, 3, and 4, (2) revise TS Limiting Condition for Operation (LCO) 3.3.4 degraded voltage relay and loss of voltage relay Limiting Safety System Settings values to reflect the revised analysis, and (3) revise the load requirement of Surveillance Requirement 3.8.1.3 to reflect values supported by the diesel generator accident loading analyses.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No

The proposed changes to LCO 3.3.2 correct an administrative error which directed inadequate action in the event that a channel of instrumentation is lost for manual safety injection initiation. The amendment places the plant in a more conservative condition, Mode 5, if the other Required Actions cannot be executed within their periodicity.

The proposed changes to LCO 3.3.4 provide setpoint changes based on a revised calculation, which generated new setpoints for the loss of voltage relays and degraded voltage relays. The new setpoints ensure the protective relays will function when required, will ensure protection from thermal damage to loads on the 480V busses, and will not cause unintended diesel generator starts even in worst case scenarios, with power provided from offsite.

The proposed changes to LCO 3.8.1 involve an increase in the minimum load band value for diesel generator surveillance SR 3.8.1.3. This change ensures that the diesel generators are capable of synchronizing with the offsite electrical system and accepting loads greater than or equal [to] the equivalent of the maximum expected accident loads. The new load band value is more conservative than the existing value and provides a more thorough test to ensure equipment emergency response capability.

Therefore, the probability or consequences of an accident previously evaluated will not be significantly increased.

2. Do the proposed amendments create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   The proposed changes involve correcting an administrative error and revising previously established values associated with the diesel generators to increase conservatism. None of these proposed changes involve a physical alteration of the plant (i.e., no new or different types of equipment will be installed) or a change in methods governing normal plant operation. The proposed changes preserve the safety analysis assumptions related to accident mitigation. No initiators or accident precursors are created by this change. Therefore, the possibility of a new or different kind of accident not previously evaluated is not created.

3. Do the proposed amendments involve a significant reduction in a margin of safety?

   Response: No

> The level of safety of facility operation is unaffected by any of the proposed changes. The requested administrative change is conservative compared to the existing requirement. The response of the diesel generators to accident transients reported in the Updated Final Safety Analysis Report (UFSAR) is unaffected by these changes. The proposed changes preserve the safety analysis assumptions related to accident mitigation. Therefore, these changes do not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Carey Fleming, Sr. Counsel – Nuclear Generation, Constellation Group, LLC, 750 East Pratt Street, 17 Floor, Baltimore, MD 21202

NRC Branch Chief: Mark G. Kowal

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

During the period since publication of the last biweekly notice, the Commission has issued the following amendments. The Commission has determined for each of these amendments that the application complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendment to Facility Operating License, Proposed No Significant Hazards Consideration Determination, and Opportunity for A Hearing in connection with these actions was published in the *Federal Register* as indicated.

Unless otherwise indicated, the Commission has determined that these amendments satisfy the criteria for categorical exclusion in accordance with 10 CFR 51.22. Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared for these amendments. If the Commission has prepared an environmental assessment under the special circumstances provision in 10 CFR 51.22(b) and has made a determination based on that assessment, it is so indicated.

For further details with respect to the action see (1) the applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment as indicated. All of these items are available for public inspection at the Commission's Public Document Room (PDR), located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management Systems (ADAMS) Public Electronic Reading Room on the internet at the NRC web site, http://www.nrc.gov/reading-rm/adams.html. If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the PDR Reference staff at 1 (800) 397-4209, (301) 415-4737 or by email to pdr.resource@nrc.gov.

Carolina Power & Light Company, Docket Nos. 50-325 and 50-324, Brunswick Steam Electric Plant, Units 1 and 2, Brunswick County, North Carolina

Date of application for amendments: July 7, 2008, as supplemented by letters dated December 17, 2008, and March 9, 2009.

Brief Description of amendments: The amendments revise Surveillance Requirement (SR) 3.6.1.6.1 to add a new requirement to verify that each vacuum breaker is closed within 6 hours following an operation that causes any of the vacuum breakers to open and, also, revise SR 3.6.1.6.2 by removing the requirement to perform functional testing of each vacuum breaker within 12 hours following an operation that causes any of the vacuum breakers to open.

Date of issuance: March 11, 2009.

Effective date: As of the date of issuance and shall be implemented within 90 days.

Amendment Nos. 251 and 279.

Facility Operating License Nos. DPR-71 and DPR-62: Amendments change the Technical Specifications.

Date of initial notice in *Federal Register*: September 23, 2008 (73 FR 54864). The supplemental letter provided clarifying information that was within the scope of the initial notice and did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 11, 2009.

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, Docket Nos. STN 50-456 and STN 50-457, Braidwood Station, Units 1 and 2, Will County, Illinois

Exelon Generation Company, LLC, Docket Nos. STN 50-454 and STN 50-455, Byron Station, Unit Nos. 1 and 2, Ogle County, Illinois

Exelon Generation Company, LLC, Docket No. 50-461, Clinton Power Station, Unit No. 1, DeWitt County, Illinois

Exelon Generation Company, LLC, Docket Nos. 50-237 and 50-249, Dresden Nuclear Power Station, Units 2 and 3, Grundy County, Illinois

Exelon Generation Company, LLC, Docket Nos. 50-373 and 50-374, LaSalle County Station, Units 1 and 2, LaSalle County, Illinois

Exelon Generation Company, LLC, Docket No. 50-352 and No. 50-353, Limerick Generating Station, Unit 1 and 2, Montgomery County, Pennsylvania

Exelon Generation Company, LLC, Docket No. 50-219, Oyster Creek Nuclear Generating Station, Ocean County, New Jersey

Exelon Generation Company, LLC, and PSEG Nuclear LLC, Docket Nos. 50-277 and 50-278, Peach Bottom Atomic Power Station, Units 2 and 3, York and Lancaster Counties, Pennsylvania

Exelon Generation Company, LLC, Docket Nos. 50-254 and 50-265, Quad Cities Nuclear Power Station, Units 1 and 2, Rock Island County, Illinois

Exelon Generation Company, LLC, Docket No. 50-289, Three Mile Island Nuclear Station, Unit 1 (TMI-1), Dauphin County, Pennsylvania

Date of application for amendments: April 21, 2008, as supplemented on March 11, 2009

Brief description of amendments: The proposed amendment removes references to and limits provided by Nuclear Regulatory Commission Generic Letter (GL) 82-12, "Nuclear Power Plant Staff Working Hours," from the subject plants' technical specifications (TS).

The references and limitations have been superseded by the requirements of Title 10 of the *Code of Federal Regulations*, Part 26 (10 CFR 26), Subpart I, "Managing Fatigue."

Date of issuance: March 23, 2009.

Effective date: As of the date of issuance and shall be implemented by October 1, 2009.

Amendment Nos.: 157, 157, 162, 162, 185, 231, 224, 192, 179, 198, 159, 274, 271, 275, 243, 238, 270.

Facility Operating License Nos. NPF-72, NPF-77, NPF-37, NPF-66, NPF- 62, DPR-19, DPR-25, NPF-11, NPF-18, NPF-39, NPF-85, DPR-16, DPR-44, DPR-56, DPR-29, DPR-30, DPR-50: The amendments revised the Technical Specifications/Licenses.

Date of initial notice in FEDERAL REGISTER: June 3, 2008 (73 FR 31721)

The March 11, 2009, supplement contained clarifying information and did not change the NRC staff's initial proposed finding of no significant hazards consideration.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 23, 2009.

No significant hazards consideration comments received: No.

Nebraska Public Power District, Docket No. 50-298, Cooper Nuclear Station, Nemaha County, Nebraska

Date of amendment request: March 24, 2008, as supplemented by letters dated September 11 and 19, 2008, November 6, 2008, and February 26, 2009.

Brief description of amendment: The amendment revised Technical Specification (TS) Section 3.7.3, "Reactor Equipment Cooling (REC) System," to allow credit for the ability to align the service water system to the REC system.

Date of issuance: March 20, 2009

Effective date: As of the date of issuance and shall be implemented within 60 days of issuance.

Amendment No.: 232

Facility Operating License No. DPR-46: Amendment revised the Facility Operating License and Technical Specifications.

Date of initial notice in *Federal Register*: April 22, 2008 (73 FR 21660). The supplemental letters dated September 11 and 19, 2008, November 6, 2008, and February 26, 2009, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register*.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 20, 2009.

No significant hazards consideration comments received: No.

Nine Mile Point Nuclear Station, LLC, Docket No. 50-220, Nine Mile Point Nuclear Station, Unit No. 1 (NMP1), Oswego County, New York

Date of application for amendment: August 15, 2008, as supplemented on December 4, 2008.

Brief description of amendments: The amendment revises NMP1 Technical Specification (TS) 6.5.7, "10 CFR 50 [Part 50 of Title 10 of the *Code of Federal Regulations*] Appendix J Testing Program Plan," to allow a one-time extension of the Integrated Leak Rate Test

(ILRT) interval for no more than 5 years. The amendment allows the next ILRT for NMP1 to be performed within 15 years from the last ILRT as opposed to the current 10-year interval.

Date of issuance: March 11, 2009

Effective date: As of the date of issuance to be implemented within 30 days.

Amendment No.: 202

Renewed Facility Operating License No. DPR-063: The amendment revises the License and TSs.

Date of initial notice in FEDERAL REGISTER: October 21, 2008 (73 FR 62566). The supplement dated December 4, 2008, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission staff's initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 11, 2009.

No significant hazards consideration comments received: No

Northern States Power Company - Minnesota, LLC, Docket No. 50-263, Monticello Nuclear Generating Plant, Wright County, Minnesota

Date of application for amendment: April 3, 2008, as supplemented on February 23, 2009

Brief description of amendment: The amendment adopted the proposed requirements regarding control room envelope habitability set forth in Technical Specifications Task Force (TSTF) change traveler TSTF-448, Revision 3. Specifically, the amendment revised the requirements in TS Section 3.7.4, "Control Room Emergency Filtration (CREF) System," adds a new TS Section 5.5.13, “Control Room Envelope Habitability Program,” and added a license condition to the operating license to implement the TS changes.

Date of issuance: March 17, 2009

Effective date: As of the date of issuance and shall be implemented by November 1, 2009.

Amendment No.: 160

Facility Operating License No. DPR-22. Amendment revised the Technical Specifications.

Date of initial notice in FEDERAL REGISTER: May 6, 2008 (73 FR 25043)

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 17, 2009.

No significant hazards consideration comments received: No.

PPL Susquehanna, LLC, Docket Nos. 50-387 and 50-388, Susquehanna Steam Electric Station, Units 1 and 2, Luzerne County, Pennsylvania

Date of application for amendments: July 31, 2008.

Brief description of amendments: The amendments changed the PPL Susquehanna, LLC (PPL) Units 1 and 2 Technical Specification 3.6.1.3 "Primary Containment Isolation Valves (PCIVs)." It revised the Secondary Containment Bypass Leakage limit in Surveillance Requirement 3.6.1.3.11 from “less than or equal to 9 standard cubic foot/feet per hour (scfh)” to “less than or equal to 15 scfh when pressurized to greater than or equal to P_a.”

Date of issuance: March 18, 2009

Effective date: As of the date of issuance and shall be implemented within 30 days from the date of issuance.

Amendment Nos.: 251 for Unit 1 and 231 for Unit 2.

Facility Operating License Nos. NPF-14 and NPF-22: The amendments revised the License and Technical Specifications.

Date of initial notice in FEDERAL REGISTER: November 18, 2008 (73 FR 68455)

The Commission's related evaluation of the amendments is contained in a Safety Evaluation (SE) dated March 18, 2009.

No significant hazards consideration comments received: No. However, comments have been received from the Commonwealth of Pennsylvania and have been addressed in the SE.

Virginia Electric and Power Company, Docket Nos. 50-338 and 50-339, North Anna Power Station, Units 1 and 2, Louisa County, Virginia

Date of application for amendment: March 19, 2008, as supplemented October 7, 2008, November 17, 2008, and December 10, 2008

Brief description of amendment: The amendments revise the technical specifications (TSs) to 1) delete TS 3.7.13, “MCR/ESGR Bottled Air System,” 2) create TS 3.3.6, “Main Control Room/Emergency Switchgear Room (MCR/ESGR) Envelope Isolation Actuation Instrumentation,” to establish the operability requirements for the MCR/ESGR envelope isolation function, and 3) incorporate TS 3.7.14, “MCR/ESGR Emergency Ventilation During Movement of Recently Irradiated Fuel Assemblies,” into TS 3.7.10, “MCR/ESGR Emergency Ventilation System.” The changes revise the TSs to be consistent with the assumptions of the current dose analysis of record, performed in accordance with Title 10 of the *Code of Federal Regulations,* Section 50.67, “Accident Source Term,” and the results of the nonpressurized MCR/ESGR envelope tracer gas testing.

Date of issuance: March 25, 2009

Effective date: As of the date of issuance and shall be implemented within 90 days from the date of issuance.

Amendment Nos.: 255/236

Renewed Facility Operating License Nos. NPF-4 and NPF-7: Amendments change the licenses and the technical specifications.

Date of initial notice in *FEDERAL REGISTER*: April 22, 2008 (73 FR 21661)

The supplements dated October 7, 2008, November 17, 2008, and December 10, 2008, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 25, 2009

No significant hazards consideration comments received: No

Dated at Rockville, Maryland, this 30th of March, 2009.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

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